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Manager of
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June 11, 2007

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D.C. 20555

Subject: **Docket Nos. 50-361, 50-362, and 72-41**
Facility and Commitment Change Reports
San Onofre Nuclear Generating Station Units 2 and 3

Dear Sir or Madam:

This letter transmits the Facility Change Reports required by 10 CFR 50.59(d)(2) and 10 CFR 72.48(d)(2) for San Onofre Nuclear Generating Station Units 2 and 3 for the period from June 16, 2005 through February 28, 2007. The report (Enclosure 1) provides a summary of the facility changes and procedure changes, including a summary of the safety evaluations performed for each change. There were no tests or experiments during this period. The scope of these reports is based on a review of plant records and all 50.59 evaluations identified for the time period above. Complete facility change documentation is available onsite. There were no 72.48 evaluations performed by Southern California Edison (SCE) during the time period above.

Enclosure 2 provides a report on commitment changes made per Nuclear Energy Institute (NEI) "Guidelines for Managing NRC Commitment Changes," NEI-99-04, Revision 0.

If you would like any additional information, please contact Ms. Linda T. Conklin at (949) 368-9443.

Sincerely,

Enclosures: As stated

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ENCLOSURE 1

SAN ONOFRE NUCLEAR GENERATING STATION

UNIT 2 AND 3

FACILITY CHANGE REPORT (FCR)

10CFR50.59 EVALUATION SUMMARIES

FOR THE PERIOD
FROM JUNE 16, 2005 THROUGH FEBRUARY 28, 2007

AR 020401107-64, Digital Upgrade of Main Turbine-Generator Digital Electro Hydraulic (DEH) Control System (also known as the Turbine Control System (TCS)).

Description:

The Units 2 and 3 Turbine Control System (TCS) at panels 2(3) L017-1/2/3/4/5 are being upgraded from analog to digital. The new systems operate on a Distributed Control System (DCS) platform based on Westinghouse/Emerson Ovation controllers, and are designed to perform all design functions as described in the UFSAR Section 10.2.

Two Ovation controllers, designated Drop 2/52 and Drop 3/53, implement the new DCS. Drop 2/52 implements the emergency trip system (ETS); Drop 3/53, Over-speed Protection Control (OPC). Since the OPC controller implements functions of the Operator Automatic (OA) mode of operation as well as over-speed protection control, it is designated as the OA/OPC controller. Converting to a digital control system eliminates equipment obsolescence issues, improves turbine control response and reliability, eliminates some manual operator actions, and reduces turbine stresses during overspeed testing.

The changes include:

1. Replacing the existing TCS and the mechanical overspeed trip (OST) analog system with digital controls.
2. Replacing the mechanical overspeed with an electronic overspeed.
3. Changing the turbine overspeed trip setpoint to a value of 107%.
4. Allowing overspeed system testing without over-speeding the turbine.
5. Adding the following automatic turbine controls in addition to the manual controls currently in use.
    a. Turbine run-up using RPM and rate.
    b. Turbine loading using megawatts.
    c. Load change using megawatts & rate.
    d. Operator initiated Turbine run back used for loss of feed pump.
    e. Adding the ability to turn off frequency control (speed feedback loop).
6. Adding Distributed Control System (DCS) Human-System Interface (HSI) displays with control capability.
7. Adding the ability to control turbine using megawatt, and/or first stage pressure feedback loops. (This is a new feature.)
8. HP stop valves will not be closed by the governing system following a load rejection.
9. Changing the Low Pressure (LP) governor valves from modulating to open/close valves.

10. Eliminating the 125 VDC power feed for panel L017 and increasing the 125 VDC power feed to panel L002 due to the addition of the open/close solenoid valve for each LP governor valve.
11. Converting signals from the following transmitters to the Plant Computer System (PCS) from 0-10 VDC to 1-5 VDC:
    a. 2JT2900 2PT2051B 2PT2051C 2PT2051D 2ZT2200H
    b. 2PT2051A 2ZT2200A 2ZT2200D 2ZT2200E

The changes introduced by Items 2 and 5-9 above are adverse. These modifications are adverse because either they change the method of controlling the design functions or the method of controlling valves that must close during an overspeed condition for the overspeed system (Item 2, 8, and 9), change manual functions to automatic (Items 5a thru 5d and 7), or because the modification makes changes that relate to human factors issues (familiarity with the new system, adequate training, modified simulator, time for operator interaction, navigating HSI screen, initiating the appropriate action to the correct system) (Item 6).

Evaluation Summary:

The evaluation analysis concluded that the changes do not introduce the possibility of a change in the frequency of an accident because either they fail to introduce any new accident initiators, new failure modes, or because existing analysis conservatively bounds accidents that may be creditably affected. The new TCS is functionally superior to the existing system through the use of a closed feedback loop, self diagnostic and anticipatory in failure determinations which decreases operator burden and the need for intervention during normal or abnormal conditions. The HSI screens have been designed on the basis of SONGS documents JS-123-101 "SONGS Control Room Design Standard for Human Factors", and JS-023-123 "Design Standard for Plant Computer Systems/Digital Control Systems Human-System Interface" which are, in turn, based on NUREG guidelines on HSI. Further, the new system conforms with the SONGS commitments to NUREG-0700, Rev. 0.

The summary of probabilities of overspeed & rotor failure (See SO23-401-M-M393) indicate that a lower probability of failure is associated with the proposed Westinghouse TCS as indicated below.

Summary of Probabilities of Overspeed and Rotor Failure:
Existing Electro-Hydraulic Governing Mechanical Overspeed Protection: 2.13E-06
Proposed Westinghouse Ovation TCS Upgrade: 1.92E-06

The proposed activities do not introduce the possibility of a change in the likelihood of a malfunction because the activities are not initiators of any malfunction and no new failure modes are introduced.

The digital upgrade does not alter any equipment, system performance, or operator actions that could affect a turbine trip or increase main steam flow. As such, the current

UFSAR accident consequence analyses remain bounded by the events currently analyzed for Mode 1 operation at 100% reactor power. The proposed activities do not introduce the possibility of a change in the consequences of a malfunction or the possibility for a malfunction of an SSC with a different result because the activities are not initiators of any malfunctions and no new failure modes are introduced. Results of malfunctions in the turbine system remain unchanged from those previously described in the UFSAR, including failure modes of the proposed system that may result in turbine trip, increased steam demand, loss of load, or affects of turbine missiles (failure to trip).

The modification does not affect any fission product barriers described in the UFSAR or are the activities related to any methods of evaluations described in the UFSAR.

AR 020701289-37, Fix Position of Condensate Return Valve 2/3FV7546 and Remove 2/3FIC-7546

Description:

The proposed modifications change the method of providing adequate flow through 2/3RE7812, which monitors the radwaste condensate return line in the Auxiliary Steam System. The radiation monitor prevents an unacceptable release in the unlikely event the condensate system became contaminated from leakage in radwaste system components.

To compensate for varying flow conditions, the current design employs a modulating valve (2/3FV7546) in the 4" discharge line (SA1312ML032) to adjust the flowrate in a parallel ¾" sample line through 2/3RE7812. The flow control components have had several failures and have become obsolete.

Acceptable sample flowrates for the condensate return alignments described in Operating Instruction SO23-2-10 can be generated by a fixed resistance in the discharge line. The modulating valve is not required. The proposed modification creates the fixed resistance by adding a hand wheel to control valve 2/3FV7546 and setting it in a permanent position. Flow tests will establish the new fixed position. Flow control components no longer needed, including flow transmitter (rotameter) 2/3FT7546 and flow controller 2/3FIC 7546, are abandoned or removed.

Evaluation Summary:

The proposed modifications improve the reliability of the Auxiliary Steam, Miscellaneous Liquid Waste, and Process Radiological Monitoring Systems. The design bases for these systems include the capabilities for collecting and containing liquids, processing the liquid for release, and monitoring to provide early radiation detection and automatic termination of discharge in the unlikely event of radiological contamination (Ref. UFSAR Sections 11.2.1.1.1 and 11.5.1.1).

The substitution of a fixed resistance for the modulating valve does not adversely affect any of these capabilities for the following reasons:

a) The capability of the fixed resistance to generate acceptable sample (radiation monitor) flowrates under varying process conditions will be demonstrated by test prior to placing the system in service.
b) The modifications do not affect the existing radiation monitor other than to ensure that there is sufficient flow through it. The radiation monitor accuracy remains unchanged.
c) The modifications do not affect the existing controls, which alarm on high radiation or low sample flow and which automatically redirect the discharge from the main condenser to a safe location (the Misc. Wastes Tank SA1901MT063 in the Radwaste Building).

It is concluded that there are no increases in the likelihood of, nor consequences to accidents, or to SSC malfunctions, previously evaluated in the UFSAR. The offsite dose due to a release of radioactive liquid is bounded by the existing UFSAR Chapter 15.7.3.2 analysis.

AR 031100614-39,

- **ECP 031100614-3: Repair of Unit 2 Pressurizer Heater Sleeves and Lower Level Instrument Nozzles**
- **ECP 031100614-30: Removal of Unit 2 Mechanical Seal Assemblies on the Pressurizer and Steam Generator Instrument Nozzles**

Description:

This change on the Unit 2 Pressurizer repairs potentially flawed pressurizer heater sleeves and lower level instrument nozzles to prevent primary coolant leakage from these sources and removes the mechanical nozzle seal assemblies installed on both pressurizer lower level instrument nozzles and two steam generator instrument nozzles. The heater sleeve repair removes the present heater, modifies the heater sleeve, and installs a replacement heater. The fillet weld used for the sleeve replacement incorporates a larger diametrical clearance. The instrument nozzle repair removes the mechanical nozzle seal assembly and the instrument root valve adaptor assembly, modifies the instrument nozzle, and replaces the instrument root valve assembly with in-kind material. The design modifies the sleeve/nozzle by installing a new partial length sleeve/nozzle that combines a new Alloy 690 partial sleeve with the existing Alloy 600 sleeve. After the installing the modified sleeve, the RCS pressure boundary is formed by the J-groove weld between the Alloy 690 heater sleeve/nozzle and the external weld pad on the pressurizer lower head, and, for the heater sleeve, by the fillet weld between the Alloy 690 heater sleeve and the heater. The original internal J-groove weld and the area of base metal near the weld that may contain flaws remain in place. They, however, are no longer considered to be RCS pressure boundary components.

Evaluation Summary:

The proposed repair/replacement activity satisfies the commitments of the applicable Relief Requests and ASME code requirements. Therefore, the proposed change is considered to provide the required pressure boundary integrity. The alloy 690 material in the partial length sleeve is less susceptible to PWSCC than Alloy 600. Evaluation of the Boric Acid Corrosion has been performed to confirm that the pressurizer structural integrity is maintained for the remaining 40-year design life of the vessel. The inner segment of the Alloy 600 sleeve and the original J-groove weld remain in place. Flaw initiation and/or growth in the remaining Alloy 600 material and/or flaws in the original J-groove weld are not a concern because this nozzle remnant does not serve a pressure boundary or structural role. An evaluation has shown that the nozzle remnant will remain stable for the life of the component. This change meets applicable NRC requirements as well as the design, material, and construction standards applicable to repair/replacement of pressurizer sleeves/nozzles. Therefore, it is concluded that prior NRC review of this modification, beyond that obtained via associated relief requests, is not required.

AR 040901531-14, Units 2(3) Digital Feedwater Controls Upgrade

Description:

The Feedwater Control Systems (FWCS) of Unit 2&3 are being upgraded from an analog to a digital control system. The new systems operate on a Distributed Control System (DCS) platform based on Westinghouse/Emerson Ovation controllers, and will continue to perform all design functions as described in the UFSAR Section 7.7.1.3. Specific changes include:

1. Installing a digital feed water control system at 2(3)L048 and 2(3)L049 that:
    a. Extends the automatic operation to a range of 3% power to full power vs. 15% to full power.
    b. Modifies the FWCS Reactor Trip Override (RTO) to a closed loop DP control.
    c. Uses the turbine first stage pressure to validate Steam Generator (SG) flow and to provide automatic DP control during RTO.
2. Installing a DCS platform in Units 2 and 3.
3. Modifying the Main Control Room Panel 2(3)CR52.
4. Adding DCS display and control capability on two Assistant Control Room Operator (ACO) desk monitors.
5. Utilizing input from the safety-related narrow range SG E089 and SG E088 level and pressure sensors into the control system for SG flow validation.

The screen determined that extending the auto operation scope to a range of 3% to 100% reactor power and changing the Human-System Interface (HSI) potentially adversely affect the UFSAR functions. The evaluation examined each adverse change against the 50.59 criteria using the methodology outlined in EPRI TR-102348, Rev. 1. Engineering Change Packages (ECPs) 040901531-3 & 4 contain evaluations for other aspects of the change such as reliability and potential software failure to support the screen.

Evaluation Summary:

Auto operation range change:

The initiating events are not determined upon controller types, range of control, or feedwater (FW) valve closure signal source, or panel controls/displays associated with this change. Further, the lowering of Auto Operation range from 15% to 3% reactor power will not change the likelihood or frequency of an occurrence of an accident or malfunction. The selection of a detection range is not an accident initiator. The malfunctions that lead to an increase or decrease of heat removal are just as likely to happen in automatic as they are in manual. Also, lowering the Auto Operation range from 15% to 3% reactor power enables earlier selection of automatic feedwater control during the approach to full power operation. New malfunctions are not created because each drop ("drop" is a term used for a data-exchange point on network) on the network incorporates redundant features, like dual processors providing primary and backup control with complete bumpless fail-over (automatic transfer of control) on a fault. Such design features enhance the system reliability.

Human System Interface change:

The initiating events are not determined upon panel controls/displays associated with this change. Also, the HSI does not create the possibility of a system failure due to human error because there are no control functions available to the operator through the HSI. The HSI is a set of data collection and presentation devices. The current HSI is not an initiator of a malfunction associated with an excess of, or loss of heat removal. The same information presented to the operator by the existing HSI will be captured and presented by the new HSI. No control functions originate from the new HSI. Control functions continue to be independent from data collection and presentation devices.

The proposed activities are not accident initiators nor do they increase the probabilities or results of any accidents or malfunctions discussed in the UFSAR. No new failure modes are introduced as a result of these activities. There are no radiological consequences associated with the control activities being modified. Therefore, these activities may proceed without seeking NRC concurrence via a license amendment.

AR 050801215-8, Changes to the U3C14 core fuel loading pattern

Description:

SONGS is changing the Unit 3 Cycle 14 core fuel loading pattern. The change will allow extending the Unit's cycle length to 600 EFPD which, in turn, allows a reduction in the length of mid-cycle outage. Additionally, the fuel cladding material used on the 100 fresh Batch R fuel assemblies is being changed to ZIRLO to improve corrosion resistance of fuel cladding. As a consequence of the new ZIRLO cladding, the methodologies used to perform non-LOCA safety analyses and the evaluation model used to perform LOCA evaluations have been modified to account for the presence of ZIRLO cladding in the core. The core fuel loading pattern change increases the fuel failure prediction for the Unit 3 Increased Main Steam Flow with Single Failure (IMSF+SF) event. The increased fuel failure increases the U3 IMSF+SF event two-hour Exclusion Area Boundary thyroid and whole body dose consequences beyond the values reported in the UFSAR. The cladding material change and the methodology change screened out.

Evaluation Summary:

The evaluation determined:

(1) The change to the core fuel loading pattern is not an initiator of any accident and no new failure modes are introduced. Hence, accident and malfunction frequencies previously evaluated remain unaffected.
(2) The activity does not introduce the possibility of a change in the consequences of any malfunction previously evaluated in the UFSAR.
(3) The activity does not create the possibility of an accident of a different type or a malfunction with a different result than any previously evaluated in the UFSAR.
(4) The activity does not exceed or alter the DBLFPBs (Design Base Limit for a Fission Product Barrier) or methods of evaluation described in the UFSAR.
(5) The radiological consequences of changing the core fuel loading pattern are no more than a 4.3 rem increase in the EAB thyroid dose and no more than a 0.7 rem increase in the EAB whole body dose. These dose increases are less than or equal to 10 percent of the difference between the current UFSAR dose values and the 10 CFR 100 Section 100.11 regulatory guideline values, and
(6) The proposed doses do not exceed the current guideline values for the IMSF+SF event.

Based on Items (5) and (6), the changing of the core fuel loading pattern results in no more than a minimal increase in the radiological consequences of the U3 IMSF+SF event previously evaluated in the UFSAR. Hence, prior regulatory approval for the change is not required.

AR 060101335-13, Reduction in Post-LOCA Containment Air Mixing Flow Rate

Description:

The proposed activities reduce the number of Dome Air Circulator (DAC) fans credited for containment sprayed and unsprayed region mixing from 2 fans to 1 fan, and reduces the flow rate for the one operational DAC fan modeled in the UFSAR Section 15.10.6.3.3 "Loss of Coolant Dose Analysis" from 37,000 cfm to 0 cfm, a conservative value. The changes are being made to align the analysis model with the bases to Tech Spec 3.6.8.

The reduction in the DAC fan flow rate contribution to the containment air mixing flow rate reduces the particulate iodine removal by the containment spray system. The increase in the leakage of particulate iodine to the outside environment is considered adverse. The reduced flow rate causes the offsite Exclusion Area Boundary (EAB) and Low Population Zone (LPZ) thyroid inhalation dose consequences to increase. The EAB and LPZ whole body gamma doses, and the control room doses, are unchanged.

Evaluation Summary:

The evaluation determined

(1) The containment air mixing flow rate change is not an initiator of any accident and no new failure modes are introduced, hence, accident and malfunction frequencies previously evaluated remains unaffected.

(2) The activity does not introduce the possibility of a change in the consequences of any malfunction previously evaluated in the UFSAR.

(3) The activity does not create the possibility of an accident of a different type or a malfunction with a different result than any previously evaluated in the UFSAR.

(4) There are no design basis limits for fission product barriers or methods of evaluation described in the UFSAR applicable to the containment air mixing flow rate change.

(5) The radiological consequences of reducing the DAC fan flow rate contribution to the containment mixing rate modeled in the Loss of Coolant Accident Dose Analyses are no more than 0.2 rem in the EAB and LPZ thyroid inhalation doses which is less than or equal to 10 percent of the difference between the current UFSAR dose values and the 10 CFR 100 Section 100.11 regulatory guideline values, and

(6) The proposed doses do not exceed the current NUREG-0800 Standard Review Plan Section 15.6.5 guideline values for the LOCA event.

Based on Items (5) and (6), the reduction in the DAC fan flow rate modeled in the LOCA dose analysis results in no more than a minimal increase in the radiological consequences of the LOCA event previously evaluated in the UFSAR.

AR 060401009-6, LCS change L06-004

Description:

The evaluation examines a one-time change to the SR 3.3.109.1 FREQUENCY from 30 days to 60 days as contained in LCS 3.3.109.1 for the high pressure turbine stop and control valves to support the return to service from the Unit 2 Cycle 14 outage. The one-time change expires May 27, 2006. All LP and HP stop and control valves were tested mid March 2006. Testing the valves again on-line prior to turbine roll with existing procedures requires closing the MSIVs, resulting in loss of vacuum and the possible need to re-warm the steam lines. Warming the steam lines with Unit 3 out of service involves blowdown of steam traps in the turbine building, which is undesirable from a personnel safety point of view. Other alternatives are not supported by existing procedures and involve the need to address risks associated with using a digital control system that has not been fully tested.

Evaluation Summary:

UFSAR section 3.5.1.3.3.1, "Probability of Missile Genesis (P1)", states the design function of the high pressure (HP) valve test is to keep missile generation probability below 1 X 10E-5. The turbine missile analysis, SO23-401-4-165, assumes monthly testing of the HP turbine valves. LCS 3.3.109.1 requires monthly HP turbine valve testing to ensure that the turbine overspeed protection instrumentation and control valves are operable and will protect the turbine from excessive overspeed. Protection from turbine excessive overspeed is required since excessive turbine overspeed potentially generates missiles that could impact and damage safety related components, equipment, or structures.

AR060401009-5 documents Alstom's assessment of the impact of changing the LCS HP valve test frequency from 31 days to 60 days on the results of the latest missile analysis and concludes that the probability of missile genesis (P1) is 4.03E-6, well below the 1 X 10E-5 acceptance criteria provided in the stated UFSAR section.

Further, per the Alstom missile analysis, the probability of a malfunction of the turbine valves increases proportionately to the increase in test interval. Increasing the test interval from 31 days to 60 days increases the probability of a malfunction of the turbine valves by less than a factor of 2.

Therefore, changing the LCS required frequency of testing the HP valves from 31 days to 60 days does not result in more than a minimal increase in frequency of occurrence of an accident or more than a minimal increase in the likelihood of occurrence of a malfunction of an SSC important to safety previously evaluated in the UFSAR.

The potential consequence of a turbine missile remains unchanged regardless of the change in probability of initiating a missile. Thus, there is no increase in the consequences of an accident or consequence of a malfunction of a SSC previously evaluated in the UFSAR.

The proposed activity only increases the probability of generating a missile and does not create a possibility for an accident of a different type or the possibility for a malfunction of a SSC important to safety with a different result than any previously evaluated in the UFSAR. Further, this change will not impact a design basis limit for a fission product barrier or result in a departure from a method of evaluation described in the UFSAR. Therefore, the change can be made without seeking regulatory permission.

AR 060800698-13, Containment Hydrogen Recombiner Removal for One Cycle

Description:

This change removes Containment Hydrogen Recombiner (E146) for one cycle of operation (Cycle 15) to facilitate Steam Generator replacement. Following Steam Generator replacement installation in the Cycle 16 RFO, SCE will restore the recombiner to its original location.

The hydrogen recombiner performs no active function in any safety analyses. Removing the component has no impact on its passive function, as a containment heat sink, and thus no impact on the Containment P-T analysis. Therefore, the proposed activity has no more than minimal effects.

Evaluation Summary:

The Hydrogen Recombiners are no longer part of the Containment Combustible Gas Control System. UFSAR Sections 6.2.5 and 15.6.3.3.5.1C states: "SONGS was granted an exemption from certain requirements of 10 CFR 50.44 and 10 CFR 50, Appendix A, General Design Criteria 41." The exemption allowed removing hydrogen control requirements from the SONGS Units 2 and 3 design bases. As a result, design bases post-accident dose calculations no longer considered dose consequences related to hydrogen purge system operation. Thus, the proposed activity has no impact to the Containment Combustible Gas Control System or post-accident dose licensing bases.

The hydrogen recombiners are identified in UFSAR Table 3.10B-1 as "Non-NSSS Seismic Category I Electrical and Instrumentation Equipment Qualification", (Sheet 2 of 14) and UFSAR Table 8.3-1 "List of Loads Supplied by Class 1E AC System" (pages 8.3-5 and 8.3-10) as receiving Class 1E electrical power. The proposed activity will provide electrical termination, removal, and storage in accordance with standard electrical system design criteria. The hydrogen recombiner's removal has no impact on Environmental or Seismic Qualification, and will not create any new failure modes.

UFSAR Table 6.2-13 (Sheet 10 of 18) identifies the hydrogen recombiners as "Passive Heat Sinks". Material identified in the table is used within the Containment Pressure-Temperature (P-T) analyses for LOCA and MSLB. The table shows the hydrogen recombiner's surface area per unit is 191 ft^2, less than 0.03% of the total containment building passive metal heat sinks - 653,186 ft^2. In addition, the containment structure passive heat sinks (concrete) account for another 598,509 ft^2. Removing one hydrogen recombiner affects the total passive heat sink surface area by less than 0.02%. Any resultant change to the post-LOCA or post-MSLB peak pressure/temperature or Equipment Qualification temperature would be imperceptible, since this heat sink input change is within the total accuracy of the calculation model.

In summary, the hydrogen recombiner is not credited to perform an active function in any safety analyses. Its passive function, as a containment heat sink, has no impact on the Containment P-T analysis. Therefore, the effect of this proposed activity is no more than minimal.

ENCLOSURE 2

SAN ONOFRE NUCLEAR GENERATING STATION

UNIT 2 AND 3

REPORT ON THE COMMITMENT CHANGES MADE PER NEI "GUIDELINES FOR MANAGING NRC COMMITMENT CHANGES"

REPORT ON COMMITMENT CHANGES MADE PER "NEI GUIDELINES FOR MANAGING NRC COMMITMENTS"

NRC letter to Mr. Ralph Beedle, Senior Vice President and Chief Nuclear Officer, Nuclear Energy Institute (NEI), dated March 31, 2000 and SECY-98-224, "Staff and Industry Activities Pertaining to the Management of Commitments Made by Power Reactor Licensees to the NRC," both state that the NEI 99-04 "Guidelines for Managing NRC Commitments Changes" Revision 0, dated August 2, 1999, was an acceptable guide for licensees to follow for managing and changing their commitments to the NRC. Part of the commitment change process, given in the NEI guidelines, identifies that various commitments can be changed with the notification to the NRC made in a report submitted annually or with the FSAR updates as required by 10CFR50.71(e). The intent of this report would be to provide a brief summary of the commitments changed since the last report in lieu of filing individual notifications as commitments are revised.

The following summarizes the commitment changes for San Onofre Units 2 and 3 that are to be reported.

1. CONTROL OF SAFEGUARDS INFORMATION (SGI) ACCESS

10CFR73.21(d)(1) and (2) require that safeguards information shall be under the control of an authorized individual or if in storage, shall be in a locked security storage container. NRC Inspection Report 50-361/95-15; 50-362/95-15, dated 8/24/95, noted five instances of failing to protect safeguards information, including two lost safeguards documents. NRC Inspection Report 50-361/97-24; 50-362/97-24, dated 12/24/97 identified an apparent violation involving the failure to adequately protect a copy of the safeguards contingency plan. This resulted in a Notice of Violation, dated 2/18/98, for the loss of a copy of the Safeguards Contingency Plan.

The reply to the notice of violation, dated 3/20/98, committed to having single point control of SGI access. This was implemented by establishing a single SGI Custodian having access to each SGI storage container. The commitment to having single point control of SGI access was made to prevent future occurrences, not to restore compliance with a regulation.

Based on historical data since 1998 and benchmarking of other facilities, the control of SGI access has been changed to allow additional SGI custodians within the same SONGS division to know the combination of a SGI container besides the SGI custodian who controls the container without the need to change the combination. This allows for flexibility while still maintaining control of the locked security storage containers. This meets the 10CFR73.21 (d)(1) and (2) requirements of SGI being under the control of an authorized individual or if in storage, shall be locked in a locked security storage container. Following the NEI Guidelines for managing NRC Commitment Changes, the original commitment has been revised to allow multiple SGI Custodians to have access to a single SGI storage container.

This commitment was made to minimize the recurrence of an adverse condition. Following the NEI Guidelines for Managing NRC Commitment Changes, this commitment can be revised and the NRC notified in the next Refueling interval summary report (Facility Change Report).

2. REVISING THE METHOLODOGY OF COMPONENT COOLING WATER (CCW) HEAT EXCHANGER TESTING

In the March 29, 1991 "Program Response for Generic Letter 89-13, Service Water System Problems Affecting Safety-Related Equipment," SCE informed the NRC that part of the program to address Service Water System problems affecting safety-related equipment would include performance testing of the CCW heat exchangers every refueling outage while on shutdown cooling at the start of refueling outages. After three tests, the licensees were allowed to determine the best test frequency to provide assurance that the equipment will perform the intended safety functions during the intervals between tests with the minimum extended frequency being 5 years. After several inspections, it was determined that the CCW heat exchanger performance tests could be conducted under normal operating conditions. This commitment change was transmitted in SCE Facility Change Report dated January 21, 1998. As a result of evaluating the effects of backflushing, it was determined that the CCW heat exchanger performance tests should be performed on only one CCW heat exchanger during each refueling outage after backflushing with the heat load at least equal to 20% of the design heat load. The other CCW heat exchanger would be tested on-line, under normal operating conditions, for information only.

This is a change to a commitment made in response to a Generic Letter which has been implemented. Following the NEI Guidelines for Managing NRC Commitment Changes, this commitment can be revised and the NRC notified in the next Refueling interval summary report (Facility Change Report).